LEARNING OBJECTIVES:

2.04.01 Identify the DOE external exposure limits for occupational workers.

2.04.02 Identify the DOE limits established for the embryo/fetus of a female occupational worker.

☞ 2.04.03 Identify the administrative exposure control guidelines at your site, including those for the:

a. Radiation worker
b. Non-radiation worker
c. Incidents and emergencies
d. Unborn child of a female occupational worker.

☞ 2.04.04 Identify the requirements for a female radiation worker who has notified her employer in writing that she is pregnant.

2.04.05 Determine the theory of operation of a thermoluminescent dosimeter (TLD).

2.04.06 Determine how a TLD reader measures the radiation dose from a TLD.

2.04.07 Identify the advantages and disadvantages of a TLD.

☞ 2.04.08 Identify the types of beta-gamma TLDs used at your site.

☞ 2.04.09 Identify the types of neutron TLDs used at your site.

☞ 2.04.10 Determine the requirements for use of TLDs used at your site.

☞ 2.04.11 Determine the principle of operation, and the types used, for the personnel neutron dosimeters used at your site.

☞ 2.04.12 Determine the principle of operation of self-reading dosimetry (SRD) used at your site.

☞ 2.04.13 Determine the principle of operation, and guidelines for use, for the alarming dosimeters used at your site.

☞ 2.04.14 List the types of bioassay monitoring methods at your site.

INTRODUCTION

Dosimetry is the quantitative assessment of radiation received by the human body. Biological effects of radiation and their prevention is of primary interest to an RCT. The use of the reddening of the skin (erythema dose of approximately 300 R) in the early 1900's as a standard of "too much" was insensitive for measuring levels related to biological effects.

DOE ADMINISTRATIVE CONTROL LEVELS AND DOSE LIMITS

The DOE's objective is to maintain personnel radiation exposure well below regulatory dose limits. To accomplish this objective, challenging numerical Administrative Control Levels are established below the regulatory limits to administratively control and help reduce individual and collective radiation dose. In general, efforts to reduce individual dose should not be allowed to cause a concurrent increase in collective dose. These control levels are multi-tiered with increasing levels of authority required to approve higher Administrative Control Levels.

A committed effective dose equivalent is used to assess internal dose received by personnel at DOE facilities. The committed effective dose equivalent is the resulting dose committed to the whole body from internally deposited radionuclides over a 50-year period after intake.

Unless otherwise indicated, administrative, lifetime and special control levels and dose limits are stated in terms of the sum of the doses received from internal and external sources.

Administrative Control Level *(RCM Article 211)*

A DOE Administrative Control Level of 2,000 mrem per year per person is established for all DOE activities. Approval by the appropriate Secretarial Officer or designee shall be required prior to allowing a person to exceed 2,000 mrem.

An annual facility Administrative Control Level shall be established by the contractor senior site executive based upon an evaluation of historical and projected radiation exposures, work load and mission. The selection of the specific value shall be more restrictive than the DOE Administrative Control Level. This control level should be reevaluated annually. The choice of a low level for one year should not preclude choosing either a higher or lower level in a subsequent year.

For most facilities, an annual facility Administrative Control Level of 500 mrem or less should be challenging and achievable. An annual Administrative Control Level above 1,500 mrem is in most cases not sufficiently challenging to meet the goals of the RCM.

No person shall be allowed to go above the facility Administrative Control Level without the prior approval of the contractor senior site executive.

Lifetime Control Level *(RCM Article 212)*

To administratively control a worker's lifetime occupational dose, a Lifetime Control Level of N rem shall be established where N is the age of the individual in years. Special Control Levels shall be established for individuals who have doses exceeding N rem.

The internal contribution to lifetime occupational dose from intakes prior to January 1, 1989, should be calculated in terms of either cumulative annual effective dose equivalent or committed effective dose equivalent. The internal contribution to lifetime occupational dose should continue to be reassessed as further bioassay results and improved methods for assessing internal dose become available.

2.04.01 Identify the DOE external exposure limits for occupational workers.

Radiological Worker Dose Limits *(RCM Article 213)*

Dose limits are provided in Table 1 and shall not be exceeded. All occupational exposure received during the current year shall be included when demonstrating compliance with dose limits. These regulatory limits are established by the "Radiation Guidance to Federal Agencies for Occupational Exposure" signed by the President.

Radiological workers from other DOE or DOE contractor facilities may receive occupational exposure as a radiological worker if they:

- Provide a record of current Radiological Worker I or II standardized core training.
- Receive site-specific Radiological Worker I or II training at the facility where they will be working.
- Provide their radiation dose records for previous years and written estimates, signed by the individual, for the current year.

Proposed use of the Planned Special Exposure as specified in 10 CFR 835 shall be applied only in extraordinary situations and when the following requirements have been met:

- The proposed activity has been reviewed by the Radiological Control Manager and submitted by the senior site executive to the lead Secretarial Officer for approval.

- The proposed activity has been jointly approved by the Secretarial Officer and the Assistant Secretary for Environment, Safety and Health.

Visitor Dose Limit *(RCM Article 214)*

Visitors to DOE sites shall be limited to an annual radiation dose of 100 mrem from the sum of internal and external radiation sources unless they qualify as radiological workers. (See Table 1)

Special Control Levels *(RCM Article 216)*

Certain situations require individualized exposure control levels. In addition to considering recommendations from senior radiological control and medical officials, the contractor senior site executive should obtain advice from other disciplines such as human resources and legal in establishing Special Control Levels. The contractor senior site executive may wish to establish these Special Control Levels using a Radiological Health Advisory Group.

A Special Control Level for annual occupational exposure shall be established for each monitored person with a lifetime occupational dose exceeding N rem, where N is the age of the individual in years. The Special Control Level shall not exceed 1 rem and should allow the individual's lifetime occupational dose to approach N rem as additional occupational exposure is received.

The employer should be attentive to special concerns of employees, such as those undergoing radiation therapy, and establish Special Control Levels as appropriate.

Table 1 - Summary of Dose Limits

TYPE OF EXPOSURE		ANNUAL LIMIT
Radiological Worker*:	Whole Body (internal + external)	5 rem
Radiological Worker*:	Lens of Eye	15 rem
Radiological Worker*:	Extremity (hands and arms below the elbow; feet and legs below the knees)	50 rem
Radiological Worker*:	Any organ or tissue (other than lens of eye) and skin	50 rem
Declared Pregnant Worker:	Embryo/Fetus	0.5 rem per gestation period
Minors and Students: Whole body (internal + external) (under age 18)		0.1 rem
Visitors** and public: Whole Body (internal + external)		0.1 rem

* Radiological workers are General Employees authorized unescorted access to radiological areas.

** Applies to visitors who have not completed training.

Notes:

- Internal dose to the whole body shall be calculated as committed effective dose equivalent. The committed effective dose equivalent is the resulting dose committed to the whole body from internally deposited radionuclides over a 50-year period after intake.
- The annual limit of exposure to "any organ or tissue" is based on the committed dose to that organ or tissue resulting from internally deposited radionuclides over a 50 year period after intake plus any external effective dose equivalent to that organ during the year.
- Exposures due to background radiation, therapeutic and diagnostic medical procedures, and voluntary participation in medical research programs shall not be included in either personnel radiation dose records or assessment of dose against the limits on this table.

2.04.02 Identify the DOE limits established for the embryo/fetus of a female occupational worker.

DOE EMBRYO/FETUS DOSE LIMITS *(RCM Article 215)*

After a female radiological worker voluntarily notifies her employer in writing that she is pregnant, for the purposes of fetal/embryo dose protection, she is considered a declared pregnant worker. This declaration may be revoked, in writing, at any time by the declared pregnant worker.

The employer shall provide the option of a mutually agreeable assignment of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure is unlikely.

For a declared pregnant worker who chooses to continue working as a radiological worker:

- The dose limit for the embryo/fetus from conception to birth (entire gestation period) is 500 mrem. (See Table 1)

- Measures shall be taken to avoid substantial variation above the uniform exposure rate necessary to meet the 500 mrem limit for the gestation. Efforts should be made to avoid exceeding 50 mrem per month to the declared pregnant worker.

- If the dose to the embryo/fetus is determined to have already exceeded 500 mrem when a worker notifies her employer of her pregnancy, the worker should not be assigned to tasks where additional occupational radiation exposure is likely during the remainder of the gestation period.

SITE ADMINISTRATIVE GUIDELINES

2.04.03 Identify the administrative exposure control guidelines at your site, including those for the:

a. Radiation worker
b. Non-radiation worker
c. Incidents and emergencies
d. Unborn child of a female occupational worker.

Radiological Workers

(Insert site specific information here)

Non-radiation Worker

(Insert site specific information here)

Exposure from Incidents or Emergencies

10 CFR PART 835 - Accidents and Emergencies (II.E.)

For emergency situations, general employees could be allowed to exceed specified dose limits. The level of exposure permitted will depend upon the severity of the emergency situation. Exposures up to 2 times the annual dose limits could be permitted to protect against property loss. Higher exposures, up to 5 times the annual dose limits or greater, could be permitted to save lives and protect public health. The DOE requires that the details of any exposure in excess of the annual dose limits be documented in the occupational exposure record of the affected employee. In addition, the incident must be investigated and the results reported to DOE. Departmental requirements for occurrence reporting and processing provide a mechanism for such investigations and reports. The employee must not be allowed to receive further exposure until approval is first obtained from the contractor management and responsible DOE field organization. Also, the employee must receive counseling from the appropriate health experts regarding the consequences of receiving additional occupational exposure that year and the affected employee must agree to return to radiological work. The operation that caused the

exposure must cease pending a finding by DOE that the conditions that caused the exposure had been eliminated.

(Insert site specific information here)

Unborn Child of a Female Occupational Worker

(Insert site specific information here)

2.04.04 Identify the requirements for a female radiation worker who has notified her employer in writing that she is pregnant.

SITE EXPOSURE REQUIREMENTS FOR THE UNBORN CHILD

(Insert site specific information here)

TYPES OF DOSIMETRY

As a result of irradiation, some solid substances undergo changes in some of their physical properties. These changes amount to storage of the energy from the radiation. Since the energy is stored, these materials can be used for dosimeters. The features that have been studied include:

- Optical density changes

  Optical density changes involve a change in the color of some types of plastics and glass. In glass, the dose range is 10^3 to 10^6 rads. The range for plastics is 10^6 to 10^9 rads.

- Radiophotoluminescence

  Radiophotoluminescence is the property of certain glasses (silver-activated phosphate glass) to store the energy from radiation until the glass is exposed to ultraviolet light, at which time the energy is released in the form of orange light. A fluorimeter is used to measure the light output.

The dose response with this method is quite good, but these types of dosimeters have not received wide acceptance in this country. They are, however, used extensively in Japan and Europe.

- Conductivity changes

  Very little has been done with the use of semiconductors for dosimetry applications. One reason for this is a low sensitivity of about 10 rads.

- Thermoluminescence

  The property of thermoluminescence of some materials is the main method used for personnel dosimeters at DOE facilities.

THEORY OF OPERATION OF THERMOLUMINESCENT DOSIMETERS (TLD)

Thermoluminescence is the ability of some materials to convert the energy from radiation to a radiation of a different wavelength, normally in the visible light range. There are two categories of thermoluminescence.

Fluorescence. This is emission of light during or immediately after irradiation (within fractions of a second) of the phosphor. This is not a particularly useful reaction for TLD use.

Phosphorescence. This is the emission of light after the irradiation period. The delay time can be from a few seconds to weeks or months. This is the principle of operation used for thermoluminescent dosimeters.

2.04.05 Determine the theory of operation of a thermoluminescent dosimeter (TLD).

TLDs use phosphorescence as their means of detection of radiation.

Electrons in some solids can exist in two energy states, called the valence band and the conduction band. The difference between the two bands is called the band gap.

Electrons in the conduction band or in the band gap have more energy than the valence band electrons.

Normally in a solid, no electrons exist in the energy states contained in the band gap. This is a "forbidden region."

In some materials, defects in the material exist or impurities are added that can trap electrons in the band gap and hold them there. These trapped electrons represent stored energy for the time that the electrons are held. (See figure 1) This energy is given up if the electron returns to the valence band.

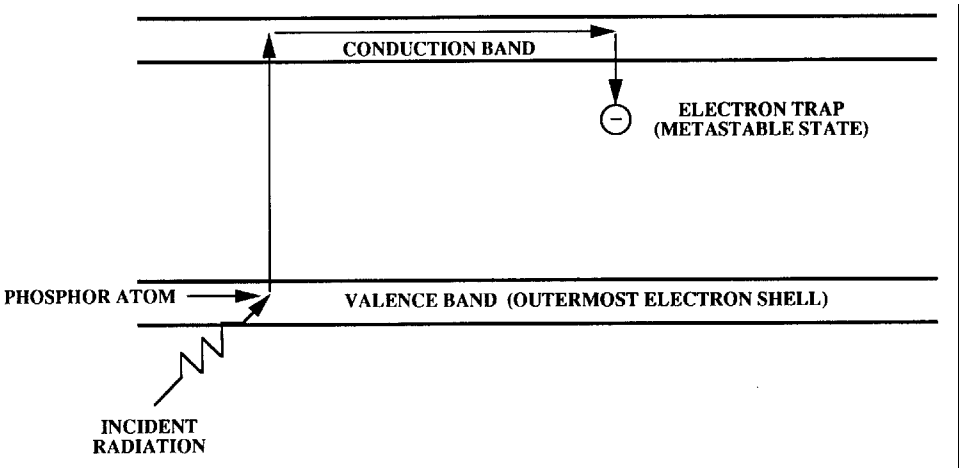

Figure 1 - Electron Entrapment

In most materials, this energy is given up as heat in the surrounding material, however, in some materials a portion of energy is emitted as light photons. This property is called luminescence. (See figure 2)

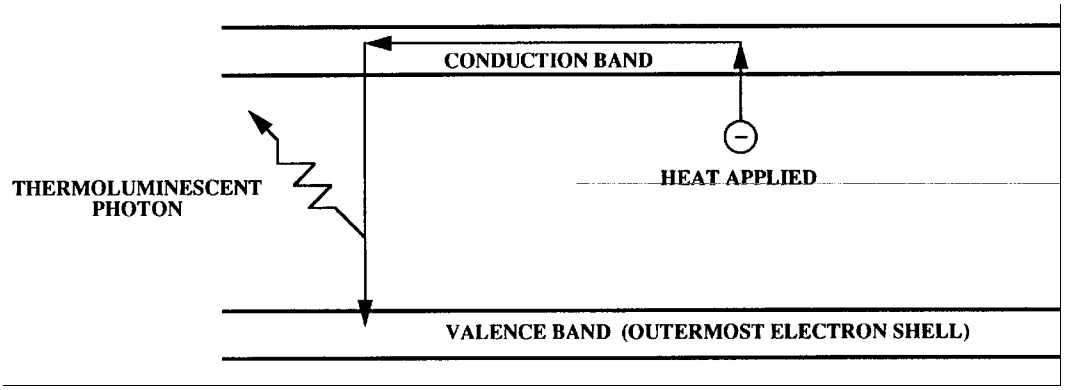

Figure 2 - Thermoluminescence

2.04.06 Determine how a TLD reader measures the radiation dose from a TLD.

OBTAINING RESULTS FROM TLDs

Heating of the TL material causes the trapped electrons to return to the valence band. When this happens, energy is emitted in the form of visible light. The light output is detected and measured by a photomultiplier tube and a dose equivalent is then calculated. A typical basic TLD reader contains the following components: (See figure 3)

- Heater - raises the phosphor temperature
- Photomultiplier Tube - measures the light output
- Meter/Recorder - display and record data

A glow curve can be obtained from the heating process. The light output from TL material is not easily interpreted. At least two peaks result. As the material is heated, electrons trapped in "shallow" traps are released. This results in a peak as these traps are emptied. The light output drops off as these traps are depleted. As heating continues, the electrons in deeper traps are released. This results in a second peak (typically larger than the first one) which is used to calculate the dose equivalent. The area under the curve is used for this calculation. (A simple glow curve is shown in figure 4.)

After the readout is complete, the TLD is annealed at a high temperature. This process essentially zeroes the TL material by releasing all trapped electrons. The TLD is then ready for reuse.

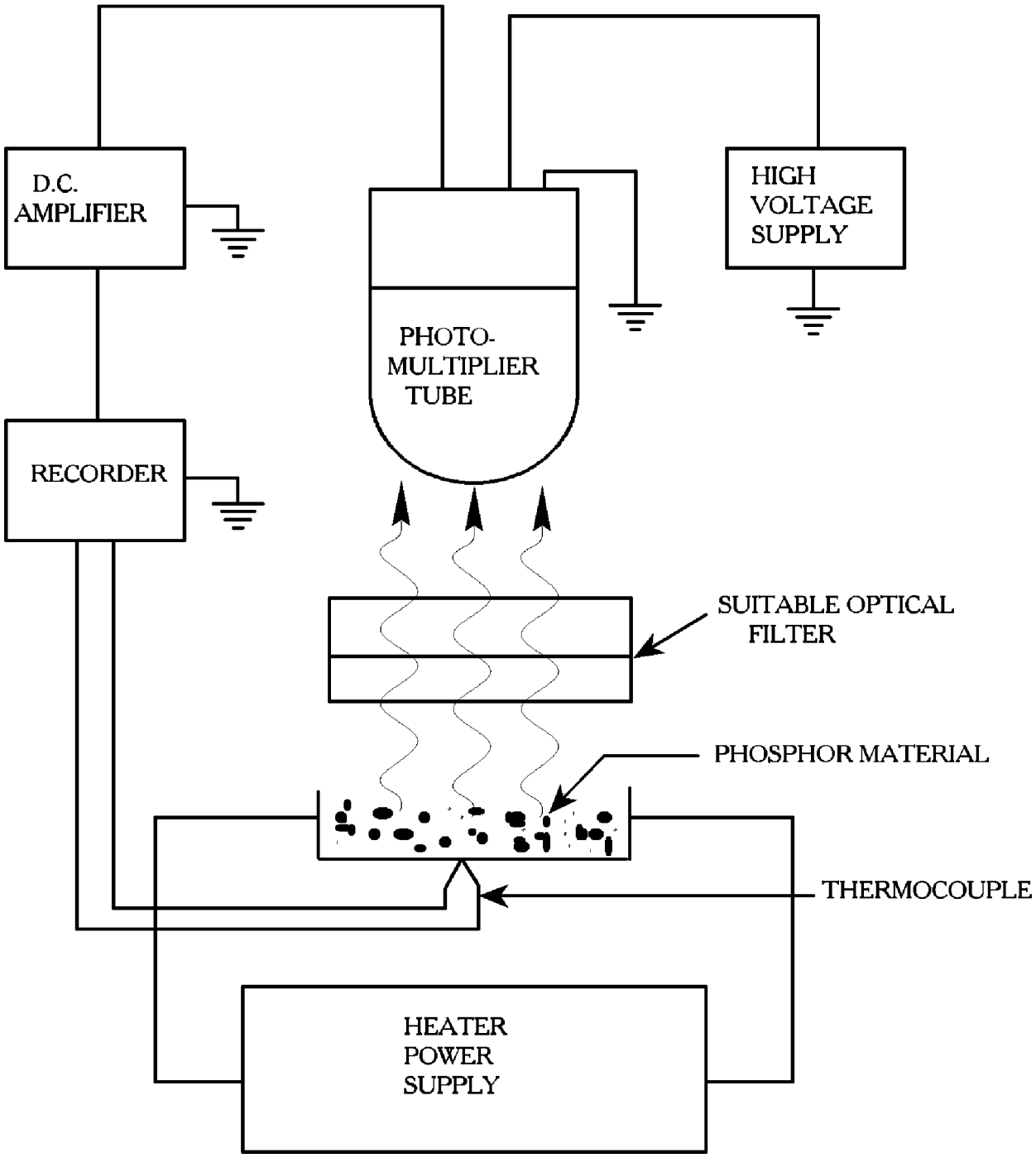

Figure 3 - TLD Reader

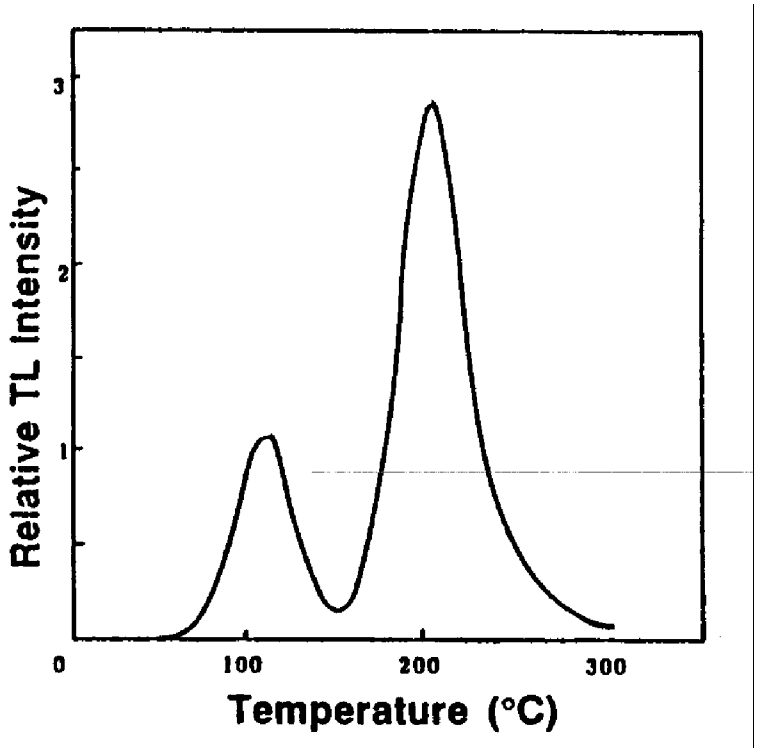

Figure 4 - Glow Curve

2.04.07 Identify the advantages and disadvantages of a TLD.

ADVANTAGES AND DISADVANTAGES OF TLDs

Advantages (primarily as compared to film badges)

- Able to measure a greater range of doses
- Doses may be easily obtained
- They can be read on site instead of being sent away for developing
- Quicker turnaround time for readout
- Reusable.

Disadvantages

- The readout process effectively "zeroes" the TLD, so the original record is lost.

2.04.08 Identify the types of beta-gamma TLDs used at your site.

SITE BETA/GAMMA TLDs

(Insert site specific material here)

2.04.09 Identify the types of neutron TLDs used at your site.

SITE NEUTRON TLDs

(Insert site specific material here)

DOE RCM EXTERNAL DOSIMETRY REQUIREMENTS *(RCM Article 511)*

Personnel dosimetry shall be required for the following:

1. Personnel who are expected to receive an annual external whole body dose greater than 100 mrem or an annual dose to the extremities, or organs and other tissues (including lens of the eye and skin) greater than 10 percent of the corresponding limits specified in Table 1

2. Declared pregnant workers who are expected to receive from external sources a dose equivalent of 50 mrem or more to the embryo/fetus during the gestation period

3. Minors and students, visitors and public expected to receive an annual external whole body dose equivalent of 50 mrem or more in a year.

Neutron dosimetry shall be provided when a person is likely to exceed 100 mrem annually from neutrons.

Dosimeters shall be issued only to personnel formally instructed in their use and shall be worn only by those to whom the dosimeters were issued.

To minimize the number of personnel in the dosimetry program, the issuance of dosimeters is discouraged to other than personnel entering Radiation Areas, High Radiation Areas or Radiological Buffer Areas where there is a potential for external exposure. Although issuing dosimeters to personnel who are not occupationally exposed to radiation can appear as a conservative practice, it creates the impression that the wearers are occupationally exposed to radiation.

Personnel shall return dosimeters for processing as scheduled or upon request, and should be restricted by line management from continued radiological work until dosimeters are returned.

Personnel shall wear their primary dosimeters on the chest area, on or between the waist and the neck, in the manner prescribed by dosimetry personnel.

Personnel shall not wear dosimeters issued by their resident facilities while being monitored by a dosimeter at another facility unless authorized by the Radiological Control Manager. Personnel shall not expose their dosimeters to security x-ray devices, excessive heat, or medical sources of radiation.

A person whose dosimeter is lost, damaged, or contaminated should place work in a safe condition, immediately exit the area and report the occurrence to the Radiological Control Organization. Reentry of the person into Radiological Buffer Areas should not be made until a review has been conducted and management has approved reentry.

2.04.10 Determine the requirements for use of TLDs used at your site.

SITE REQUIREMENTS FOR USE OF TLDs

(Insert site specific material here)

2.04.11 Determine the principle of operation, and the types used, for the personnel neutron dosimeters used at your site.

SITE PERSONNEL NEUTRON DOSIMETERS

(Insert site specific material here)

POCKET AND ELECTRONIC DOSIMETERS *(RCM Article 513)*

Pocket and electronic dosimeters are supplemental dosimeters that provide real-time indication of exposure to radiation and assist in maintaining personnel doses less than Administrative Control Levels.

Supplemental dosimeters shall be issued to personnel prior to entry into a High Radiation or Very High Radiation Area; when a person could exceed 10 percent of an Administrative Control Level from external radiation in 1 work day; or when required by a Radiological Work Permit. Pocket dosimeters should be selected with the lowest range applicable (typically 0-200 mR) for anticipated personnel exposures.

Supplemental dosimeters shall be worn simultaneously with the primary dosimeter and located on the chest area, on or between the waist and the neck.

Supplemental dosimeters shall be read periodically while in use and should not be allowed to exceed 75 percent of full scale.

Work authorized by a Radiological Work Permit shall be stopped when supplemental dosimeter readings indicate total exposure or rate of exposure substantially greater than planned. The Radiological Control Organization shall be consulted prior to continuation of work.

The energy dependence of supplemental dosimeters, particularly to low-energy beta radiation, should be considered in determining their applicability. For example, the SRPD (shown in Figure 5) has a thick case that effectively shields most betas.

Use of electronic dosimeters is encouraged for entry into High Radiation Areas or when planned doses greater than 100 mrem in 1 work day are expected. An electronic

dosimeter provides an early warning of elevated exposure through the use of alarm set points at specified dose rates or integrated doses.

When the dose results from the pocket or electronic dosimeters differ by more than 50 percent from the primary dosimeter result and the primary dosimeter result is greater than 100 mrem, an investigation should be initiated to explain the difference.

2.04.12 Determine the principle of operation of self-reading dosimetry (SRD) used at your site.

SITE SELF-READING DOSIMETERS

(Insert site specific material here)

Self Reading Pocket Dosimeters (SRPD)

The direct reading pocket dosimeter consists of an ionization chamber sensitive to a desired radiation; a quartz fiber electrometer to measure the charge; and a microscope to read the fiber image off a scale (reticle). (See figure 5)

The electrometer embodies two electrodes, one of which is a moveable quartz fiber and the other a metal frame. When the electrometer is charged to a predetermined voltage, the electrodes assume a calibrated separation.

As the dosimeter is exposed to radiation, ionization occurs in the surrounding chamber decreasing the charge on the electrode in proportion to the exposure. The deflection of the moveable quartz fiber electrode is projected by a light source through an objective lens to a calibrated scale and read through a microscope eyepiece. (See Figure 6)

Illumination for the optical system is obtained by pointing the dosimeter at any convenient light source. Light passes through the clear glass bottom seal to illuminate the scale.

The bottom is sealed by a bellows containing an insulated charging pin. When charging, the charging pin moves up to contact the electrometer closing the circuit. Sufficient voltage is applied to recharge the system. The entire dosimeter system is hermetically sealed in a protective barrel.

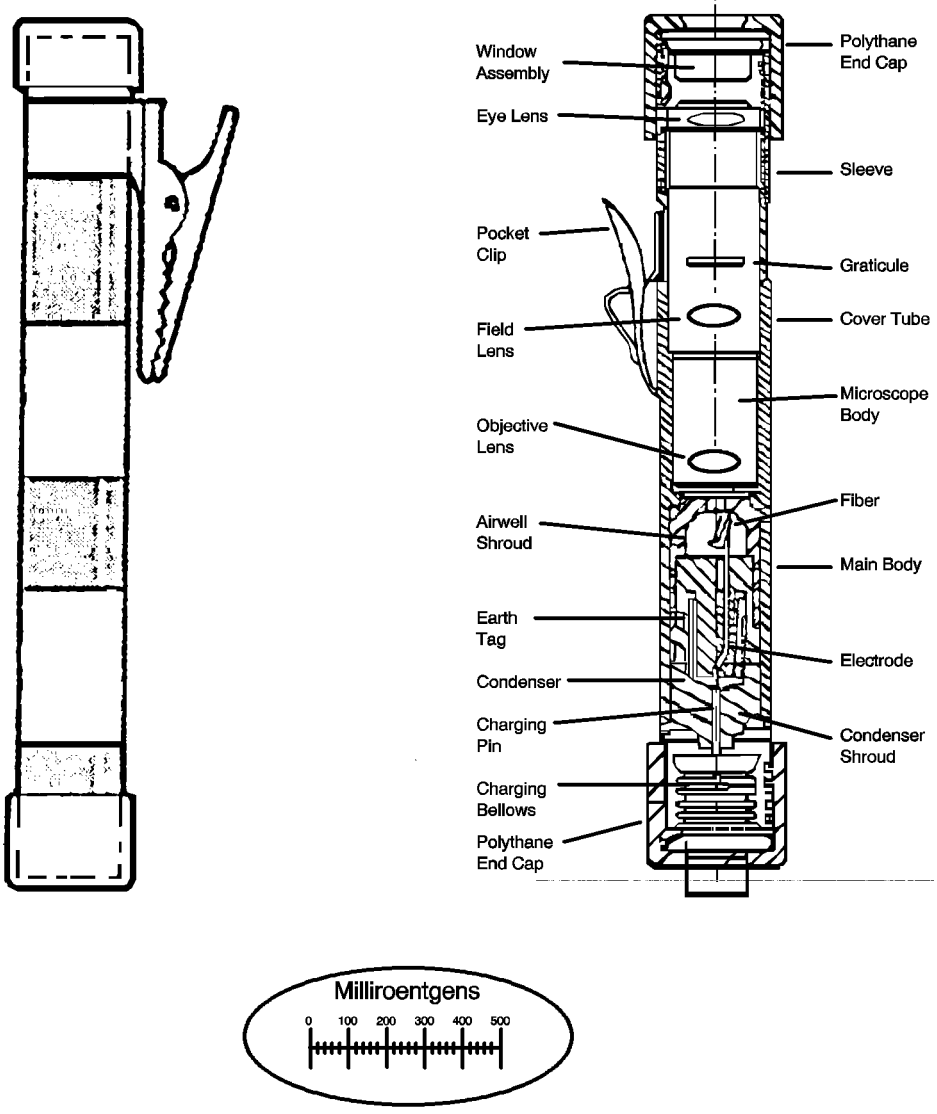

Figure 5 - SRPD

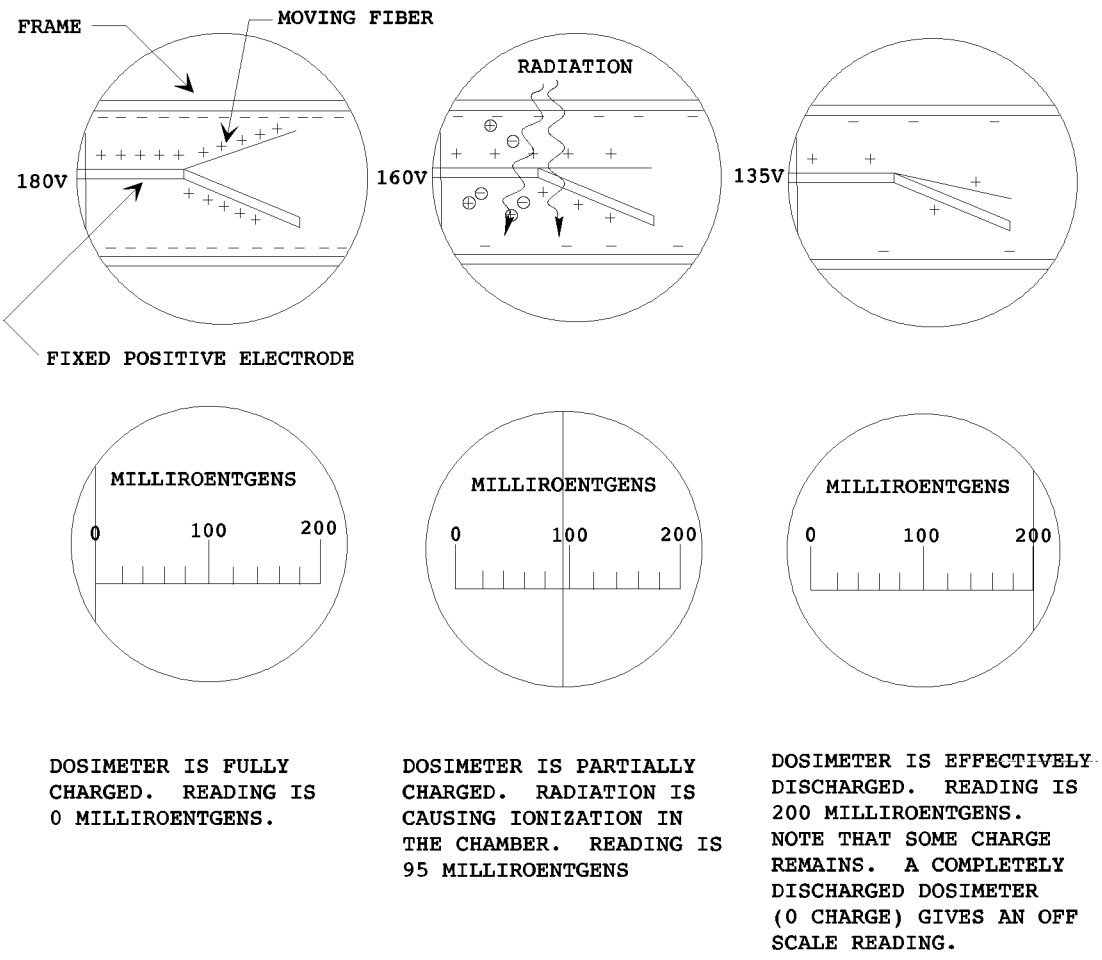

Figure 6 - SRPD Reading

SITE ALARMING DOSIMETRY

2.04.13 *Determine the principle of operation, and guidelines for use, for the alarming dosimeters used at your site.*

(Insert site specific material here)

DOE RCM INTERNAL DOSIMETRY REQUIREMENTS *(RCM Article 521)*

The following personnel shall participate in an internal dosimetry program:

- Personnel entering Radiological Buffer Areas who have the potential to receive intakes resulting in a committed effective dose equivalent of 100 mrem or more in a year
- Declared pregnant workers likely to receive intakes resulting in a dose equivalent to the embryo/fetus of 50 mrem or more during the gestation period
- Minors and students, visitors and public likely to receive intakes resulting in a committed effective dose equivalent of 50 mrem or more in a year.

The estimation of internal dose shall be based on bioassay data rather than air concentration values unless bioassay data are unavailable, inadequate, or internal dose estimates based on representative air concentration values are demonstrated to be as or more accurate.

Personnel shall participate in follow-up bioassay monitoring when their routine bioassay results indicate an intake in the current year with a committed effective dose equivalent of 100 mrem or more.

Personnel whose routine duties may involve exposure to surface or airborne contamination or to radionuclides readily absorbed through the skin, such as tritium, should be considered for participation in the bioassay program.

Personnel shall submit bioassay samples, such as urine or fecal samples, and participate in bioassay monitoring, such as whole body or lung counting, at the frequency required by the bioassay program.

Personnel shall be notified promptly of positive bioassay results and the results of dose assessments and subsequent refinements. Dose assessment results shall be provided in terms of rem or mrem.

BIOASSAY ASSESSMENT METHODS

Today's technology has not produced a device that allows accurate determination of internal exposure following the entry of radioactive materials into the body.

The method that is used to determine internal dose contributions relies on calculation of dose to affected portions of the body based on the quantities of radioactive materials in the body. Thus, the real problem becomes one of quantifying the amount of material present.

Bioassay is the term that is used to describe the assessment of the quantity of radioactive material present in the body. There are currently two types of bioassay measurements employed in nuclear industries: in vivo and in vitro. In vivo bioassay involves counting the living tissue, as described below. In vitro involves counting an excreted sample, such as urine.

Bioassay programs are designed to fulfill two needs:

1) Evaluate effectiveness of contamination control practices

- Routine bioassay programs utilize submission and analysis of samples from workers in facilities where the likelihood of intake exists
- Primarily limited to urinalysis due to ease of sample collection
- Also includes initial, routine, and termination whole body counts

2) Evaluate potential consequences of accidental inhalation or ingestion of large quantities of radioactive materials

- Can involve all types of bioassay measurements with collection and analysis of nasal, urine, and fecal samples.
- Whole body counts provide immediate indications for given isotopes if individual(s) involved are free of contamination.

Quantification of materials actually in the body can be affected by the availability of measurements taken early after the incident. The elimination rate of some materials from the body falls off as the concentration in the body falls off, or with time. Accurate quantification of initial quantities, present, thus accurate dose assessment, can be dependent on availability of early data.

Identification of the proper bioassay technique to use is aided by a knowledge of the types of contamination present in a particular work area. For example, if you know that the contamination in a facility typically includes isotopes that cannot be detected with in vivo measurements, then it would be obvious that collection and measurement of urine or other samples is necessary.

If the presence of gamma emitting nuclides is identified, consider the possibility of the presence of materials that do not decay with gamma emission. Periodic isotope assessment of contamination in facilities will provide information on relative isotope concentrations. Caution must be exercised in using information of this nature. Cycles of contamination should be used as an indicator only. Remember, fresh coolant does not have the same isotopic makeup of coolant that has decayed.

Contamination control measures cannot be too stringent during collection, handling, and analysis of bioassay samples. Cross-contamination can cause erroneous assumptions and inaccurate dose assessments. If procedural guidance is not sufficient to determine required actions, consult supervision.

IN VIVO MEASUREMENTS

In vivo techniques consist of direct measurements of gamma or X-radiation emanating from the body. This method is very useful for any radionuclide which emits (or has daughters which emit) photons of sufficient energy to escape the body. The photon flux density must be large enough for measurement in a reasonable time period, even though the quantity of material in the organ is very small.

This method is possible only for those radionuclides emitting penetrating radiation, e.g., Co-60 and Cs-137 or bremsstrahlung, e.g., P-32 and Sr-90. Many radionuclides, Na-22, Fe-59, Co-60, Zn-65, Rb-86, Sr-85, Te-132, I-131, Cs-137, Ba-140, Ce-144, Au-198, U-235, Np-239, and Am-241 emit electromagnetic radiation of sufficient energy to be measured by external counting. If the counter has been calibrated previously, one may rapidly determine the identity and amount of any of these radionuclides. Such measurements are more acceptable to the subject than the provision of samples of excreta, although they do require him to be absent from work during the period of measurement. Direct counting of the individual without preparation beforehand (changing into clean clothes and external decontamination) may give misleading results, since this method measures all gamma emitting radionuclides in or on a subject; therefore, sensitive counts (lung) should be done immediately after the subject washes and changes into clean clothing. Radon daughters that cling to body hair due to their electrostatic charge are the chief source of bad lung counts. When this method errs, it usually does so by being too high, so that a negative result is likely to be a reliable indication that there is no internal contamination with gamma emitters.

In external counting, the requirement for sensitivity and energy discrimination determines the complexity of the measuring equipment. Estimations of very small quantities require elaborate shielding of both the sensing element and the subject, sensitive detectors, and the best discrimination between gamma ray energies. However, a single moderately large, well-shielded sodium iodide crystal coupled with a multichannel analyzer can usually meet

the need. This system in conjunction with a shielded chair or moving bed, is capable of determining:

- I-131 in the thyroid gland.
- Insoluble radionuclides in the chest.
- Insoluble radionuclides in the intestine.
- Insoluble radionuclides in wounds.

These need not emit highly penetrating radiation, since much of the material may be on or near the surface, i.e., for wounds.

Because large sodium iodide crystals do not have good collimation capabilities, it is usually not possible to measure specific organ contents directly. In some cases, solid state detector (Geli) can be used for specific organ determination. However, the decreased sensitivity of this method limits the usefulness of these measurements. Small sodium iodide detectors are used for determining thyroid dose.

2.04.14 List the types of bioassay monitoring methods at your site.

Site In Vivo Methods

(Insert site specific material here)

Advantages of In Vivo Measurements

- No sample required
- Results obtained quickly
- Some equipment design allows field use
- Time and manpower requirements minimized.

Disadvantages of In Vivo Measurements

- Limited to detection and measurement of gamma emitters
- Individual must be free of contamination
- Long count times for identification
- Effects of background
- Complex calibration procedure and calibration equipment
- Expense

- Quantification error due to differences in tissue structure from one person to another as compared to calibration phantom.

IN VITRO MEASUREMENTS

The amount of material present in the body is estimated using the amount of materials present in excretions or secretions from the body. Samples could include urine, blood, breath, sputum, sweat, saliva, hair, nasal discharges, tissue and feces. Calculation requires knowledge and use of metabolic models which allow sample activity to be related to activity present in the body.

Resulting dose calculations to quantify committed and effective dose equivalents are estimates. This is due partly to use of default values for measurements that cannot be readily made such as mass of particular organs, volumes of particular fluids, etc., in lieu of actual values for individual involved. Remember that reference man is an average. Another contributing factor is the difference in metabolism from one individual to another.

2.04.14 List the types of bioassay monitoring methods at your facility.

Site In Vitro Methods

(Insert site specific material here)

Advantages of In Vitro Measurements

- Can be used for estimation of neutron doses using activation product concentration in hair and blood (P^{32} and Na^{24})
- Can be used to quantify presence of materials which decay by alpha and beta emission to allow detection and measurement with external detector systems.

Disadvantages of In Vitro Measurements

- Requires sample submission and analysis
- Time and manpower requirements

BIOASSAY SCHEDULING PROGRAM

Contamination found in a given facility will depend on the materials that are used and produced in the facility. Thus, the materials that internal dosimetrists are primarily concerned with will change from one site to another as well.

Baseline/Routine/Exit Evaluations

(Insert site specific material here)

Special Evaluations

(Insert site specific material here)

Investigation Levels

(Insert site specific material here)

Medical Uses

(Insert site specific material here)

SUMMARY

The method of operation of dosimeters is a vital knowledge for RCT. RC personnel are the first line of defense against abuse of these instruments and must ensure the proper wearing and use of them.

Internal exposure involves a source (contaminant) inside the body. It is more difficult to measure; sophisticated whole body counters or indirect measurements of excreta samples are required to obtain an estimate. The exposure from the contaminant does not stop when the person leaves the radiation field and the contaminant continues to irradiate tissue all day and all night. If necessary, medical treatment is required to enhance the removal of the source material from the body. Alpha radiation poses the biggest problem.

REFERENCES

1. "Basic Radiation Protection Technology"; Gollnick, Daniel; Pacific Radiation Press; 1983

2. **ANL-88-26** (1988) "Operational Health Physics Training"; Moe, Harold; Argonne National Laboratory, Chicago

3. **DOE/EH-0256T Revision 1** (April 1994) "U.S. Department of Energy Radiological Control Manual"

4. **10 CFR Part 835** (December 14, 1993) "Occupational Radiation Protection; Final Rule"; Federal Register; Vol. 58, No. 238